2. JENDL-3.3 Thermal Reactor Benchmark Test

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Integral tests of JENDL-3.2 nuclear data library have been carried out by Reactor Integral Test WG of Japanese Nuclear Data Committee. The most important problem in the thermal reactor benchmark testing was the overestimation of the multiplication factor of the U fueled cores. With several revisions of the data of ^{235}U and the other nuclides, JENDL-3.3 data library gives a good estimation of multiplication factors both for U and Pu fueled thermal reactors.

1. Introduction

After the release of JENDL-3.2 nuclear data library[1], Reactor Integral Test WG of Japanese Nuclear Data Committee has carried out benchmark tests of JENDL-3.2 for the fission reactor use[2,3]. In the thermal reactor benchmark tests, several problems of JENDL-3.2 were pointed out even for the most commonly used UO_2 fueled light water reactors. The new revision of JENDL-3 (revision 3) is now evaluated, and the integral tests are being carried out. In this paper, some results of JENDL-3.2 thermal reactor benchmark tests are shown, and are compared with the recent JENDL-3.3 benchmarks.

2. Benchmark cores

Thermal reactor benchmark cores include uranium, both low and medium enrichments, and plutonium fuels of oxide, metal, nitrate and silicide in pin, plate and solution form. The main benchmark cores adopted in Reactor Integral Test WG are as follows:

- TCA U cores (TCA 1.50U ~ 3.00U)
  The cores consist of light water moderated square lattice of 2.6wt.% enriched UO_2 fuel pin. The water/fuel volume ratio ranges from 1.50 to 3.00.
- TCA Pu cores (TCA 2.42PU ~ 5.55PU)
  The cores consist of light water moderated square lattice of 3.0wt.% Pu enriched UO_2-PuO_2 mixed oxide fuel pin. The water/fuel volume ratio is from 2.42 to 5.55.
- TRX cores (TRX-1, TRX-2)
  The cores consist of light water moderated triangular lattice of 1.3wt.% enriched metal U fuel pin. The water/fuel volume ratios are 2.35(TRX-1) and 4.02(TRX-2).
- STACY
  The criticality safety experiment core with 10wt.% enriched U uranyl nitrate solution.
- JRR-4
  Research reactor with light water moderated 20wt.% enriched U-silicide fuel plate.

3. Calculation code

There are used two different types of calculation code in the benchmark test. One is the continuous energy Monte Carlo code, and the other is deterministic calculation code.

The continuous energy Monte Carlo method has only few approximation in the expression of geometry and energy dependence. The point-wise cross section is directly used in the calculation, and therefore the continuous energy Monte Carlo method can be said to be most suitable for the benchmark of cross section itself. Though it usually takes a long calculation time to obtain a sufficient statistics, the Monte Carlo calculation is becoming the principal benchmark method.

On the other hand in the calculations by the deterministic method, the cross section is grouped into group constants, and geometry is also simplified. Thus it sometimes becomes not only a benchmark of cross section data, but also a benchmark of methods. Yet, there are still advantages of deterministic calculations, for example in such calculations as sample worths and reaction rate ratios and their space dependencies, which are difficult for Monte Carlo methods to reduce statistical errors. In addition, sensitivity of a core physics parameter to cross section data can be studied easier with the deterministic methods.

In the JENDL-3 thermal reactor benchmark calculations, MVP code[4] is used as a continuous energy Monte Carlo method, and as a deterministic code SRAC[5] is mainly utilized.

4. Benchmark test of JENDL-3.2

In the thermal reactor benchmark tests of JENDL-3.2, it was found that the multiplication factors of U fueled cores are overestimated, while the multiplication factors of Pu fueled cores are slightly underestimated. Figure 1 shows the MVP calculation/experiment (C/E) values of multiplication factors of U fueled cores obtained with JENDL-3.2 in comparison with those obtained with ENDF/B-VI.2. As shown in this figure, except for TRX cores, ENDF/V-VI.2 data give better estimation than that of JENDL-3.2. The TRX cores are older experiments than the other cores, and the multiplication factors of TRX cores are always underestimated with different nuclear data and calculation methods. It was therefore assumed that ENDF/B-VI.2 estimates the multiplication factor of U fueled thermal reactors better than JENDL-3.2.

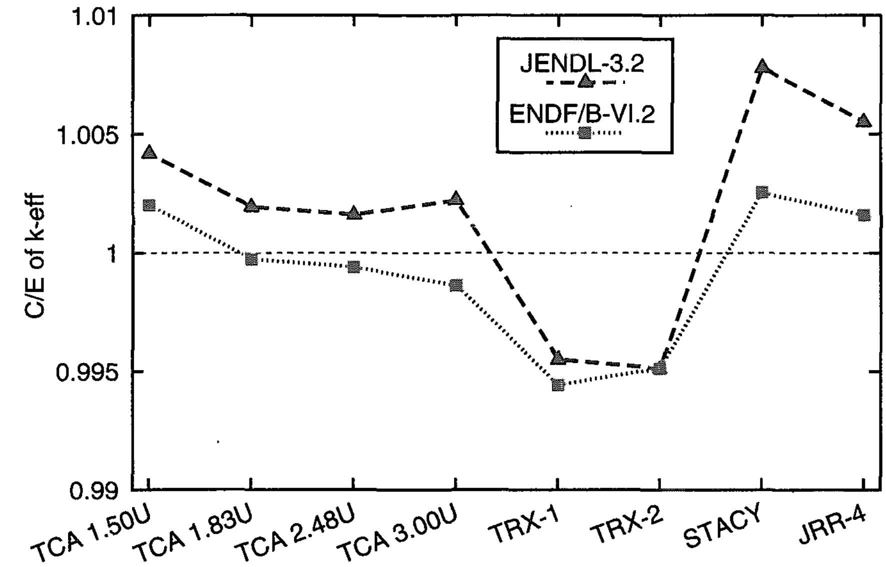

Fig. 1 C/E value of effective multiplication factors of U cores calculated with continuous energy Monte Carlo code MVP

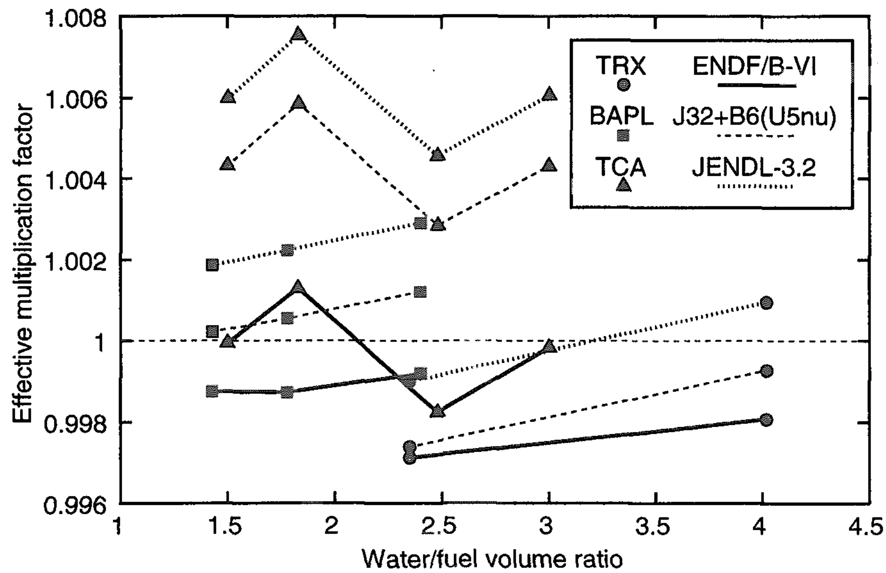

Fig. 2 Effect of ^{235}U ν value difference on multiplication factor of U cores estimated with SRAC code (J32 : JENDL-3.2, B6 : ENDF/B-VI.2, U5nu : ^{235}U ν value in thermal range)

By using the SRAC system, the effect of the difference between JENDL-3.2 and ENDF/B-VI.2 on the multiplication factor of U fueled cores was studied. As a result, the data such as ν value of ^{235}U in thermal energy range and the inelastic scattering cross section of ^{238}U were found to be different between JENDL-3.2 and ENDF/B-VI.2 libraries, and may cause the difference in the multiplication factor.

Calculated multiplication factors for some U fueled thermal benchmark cores are compared in Fig. 2 between different nuclear data libraries. In all cases, the multiplication factors obtained with JENDL-3.2 is 0.2 - 0.6% larger than that with ENDF/B-VI.2. When the ν value of ^{235}U in thermal energy range in the JENDL-3.2 based SRAC library is replaced by that of ENDF/B-VI.2, the multiplication factor decreases by about 0.2% in the all core.

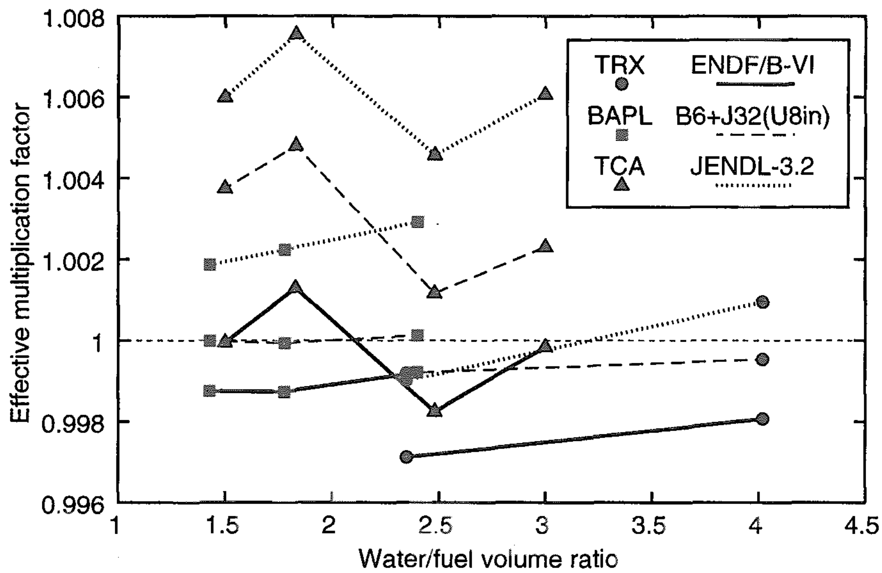

Fig. 3 Multiplication factor of U cores and the effect of the replacement of ^{238}U inelastic scattering cross section(U8in)

In the same manner as in Fig. 2, the effect of the replacement of ^{238}U inelastic scattering cross section is shown in Fig. 3. In this figure, the inelastic scattering data in the ENDF/B-VI.2 based SRAC library are replaced by those of JENDL-3.2. The effect is 0.1 - 0.4% dk and particularly large in TCA cores, of which the core volume is smaller than the other cores. It is obvious that the effect is due to the change in neutron leakage in fast energy range.

Table 1 Calculation/experiment value of spectral indices in TRX cores

core	parameter	ENDF/B-VI	B6+J32(U8in)	JENDL-3.2
TRX-1	ρ-28	1.0138	1.0118	1.0123
	δ-25	0.9925	0.9902	0.9873
	δ-28	1.0597	1.0282	1.0127
	C*	0.9956	0.9947	0.9976
TRX-2	ρ-28	0.9971	0.9960	0.9965
	δ-25	0.9788	0.9773	0.9745
	δ-28	1.0264	1.0044	0.9897
	C*	0.9890	0.9886	0.9909

ρ-28 : epithermal/thermal capture rate ratio of ^{238}U
δ-25 : epithermal/thermal fission rate ratio of ^{235}U
δ-28 : fission rate ratio of ^{238}U/^{235}U
C* : ratio of ^{238}U capture/^{235}U fission
B6 : ENDF/B-VI.2, U8in : ^{238}U inelastic scattering cross section

The effect of the inelastic scattering cross section of ^{238}U is also found in spectral indices. Table 1 summarizes the spectral indices in TRX cores. As can be seen in this table, there is a large difference in the fast neutron spectral index δ-28 between the results with JENDL-3.2 and ENDF/B-VI.2. The index denotes that the calculation with ENDF/B-VI.2 gives a harder neutron spectrum than in the JENDL-3.2 case. The difference in the spectral index δ-28 becomes much smaller when only the inelastic scattering cross section of ^{238}U is replaced.

The effects of the ν value in thermal energy region and the ^{238}U inelastic scattering cross section are both 0.1 - 0.2% dk in Pu fueled cores, and less than the case in the U cores. The effects are shown in Figs. 4 and 5, respectively.

In addition to the comparison with ENDF/B-VI.2, the comparison of JENDL-3.2 with ENDF/B-VI.5 is also performed recently. From the comparison, it is found the difference in the resonance parameter and the fission spectrum of ^{235}U both have 0.2-0.5% effect on the multiplication factors of U cores.

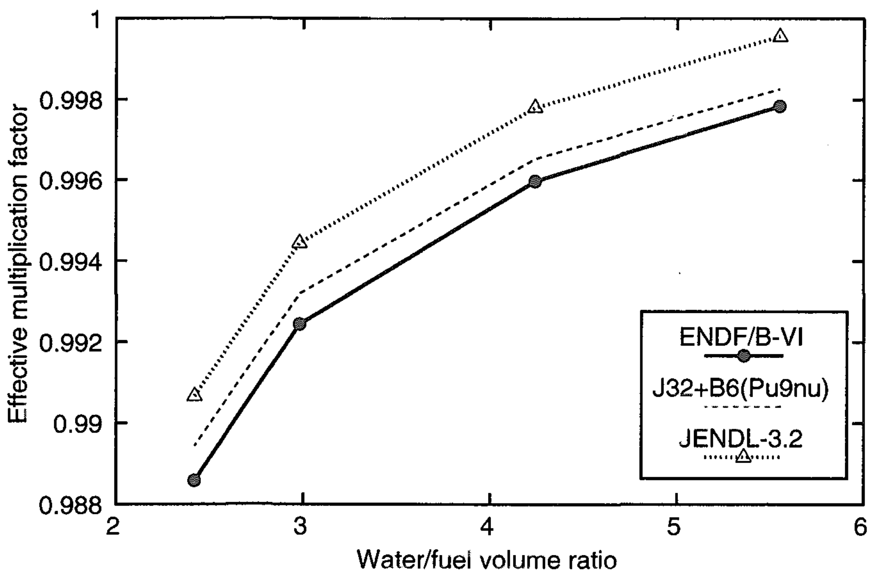

Fig. 4 Effect of Pu-239 ν value difference on multiplication factor(SRAC calculation)

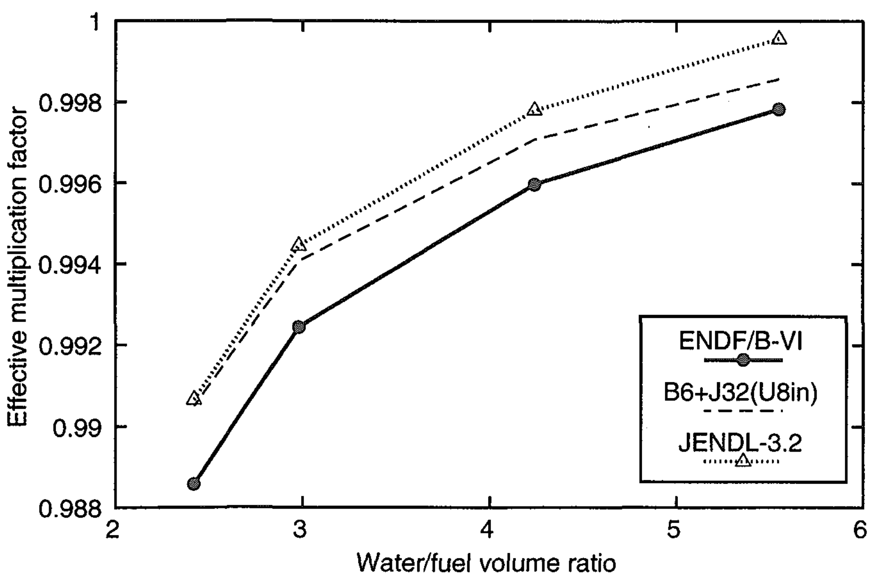

Fig. 5 Effect of U-238 inelastic scattering cross section difference on k_{eff} (SRAC calculation)

5. Thermal reactor benchmark test of JENDL-3.3

The revision of JENDL-3 nuclear data, including the ^{235}U resonance parameter and fission spectrum, has been carried out, and the new JENDL-3.3 data library is evaluated. Thermal reactor benchmark tests of JENDL-3.3 show smaller multiplication factors than those with JENDL-3.2 for U cores, and slightly larger multiplication factors than JENDL-3.2 for Pu cores. As a result, as shown in Fig. 6, the multiplication factors obtained with JENDL-3.3 are all within ±0.5% from experimental data, except for TRX cores.

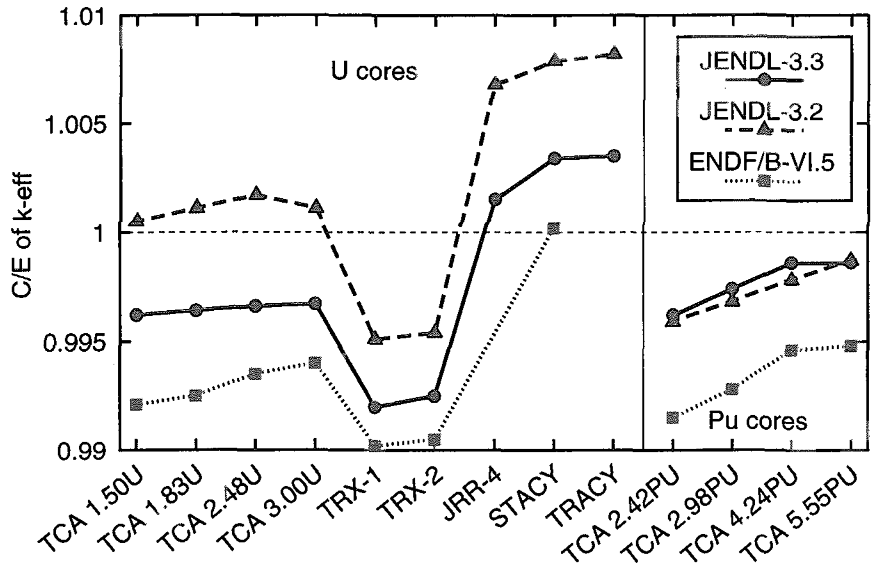

Fig. 6 Calculation/experiment value of multiplication factors of thermal benchmark cores with JENDL-3.3

6. Conclusion

With several revisions of JENDL-3.2 nuclear data library, JENDL-3.3 seems to become a reliable library for the estimation of multiplication factors both for U and Pu fueled thermal reactors. The agreement with experiment is better than those with JENDL-3.2 and ENDF/B-VI.5. Further benchmark test is necessary on reaction rate ratios, reactivity coefficients etc. to validate the reliability of JENDL-3.3 in more detail.

References

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